UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 23, 2020

Mr. James Barstow
Vice President, Nuclear Regulatory
Affairs and Support Services
Tennessee Valley Authority
Sequoyah Nuclear Plant
1101 Market Street, LP 4A
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNIT 2 - ISSUANCE OF EXIGENT AMENDMENT NO. 342 TO OPERATE ONE CYCLE WITH ONE CONTROL ROD REMOVED (EPID L-2020-LLA-0078)

Dear Mr. Barstow:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 342 to Renewed Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Unit 2. The amendment consists of changes to the technical specifications in response to your application dated April 17, 2020, as supplemented by a letter dated April 22, 2020.

The amendment revises the Sequoyah Nuclear Plant, Unit 2, Technical Specification 4.2.2, "Control Rod Assemblies," to allow Unit 2 to operate for Operating Cycle 24 with 52 full-length control rod assemblies instead of 53 full-length assemblies.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Perry H. Buckberg, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-328

Enclosures:
1. Amendment No. 342 to DPR-79
2. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 342
Renewed License No. DPR-79

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Tennessee Valley Authority (the licensee), dated April 17, 2020, as supplemented by a letter dated April 22, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-79 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 342 are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 24 hours from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and
Technical Specifications

Date of Issuance: April 23, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 342

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-79

SEQUOYAH NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-328

Replace the following pages of Renewed Facility Operating License No. DPR-79 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3	3
13b	13b

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain a marginal line indicating the area of change.

REMOVE	INSERT
4.0-1	4.0-1
4.0-2	4.0-2

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 342 are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Initial Test Program</u>

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the criteria in the endorsed ASME/ANS RA-Sa-2009 PRA Standard for other external hazard screening significance; as specified in Unit 2 License Amendment 340.

(2) Prior to implementation of the provisions of 10 CFR 50.69, TVA shall complete the items below;

a. Items listed in Attachment 1, "SQN 10 CFR 50.69 PRA Implementation Items," in TVA letter CNL-19-002, "Response to Request for Additional Information Regarding Application to Modify Sequoyah Nuclear Plant Units 1 and 2, Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors, (SQN-TS-17-06) (EPID: L-2018-LLA-0066)," dated March 21, 2019.

(3) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach, change from alternative method for internal fire to a fire probabilistic risk assessment approach).

(28) Prior to Cycle 24 startup from Unit 2 Refueling Outage 23, TVA shall ensure the Cycle 24 core design will not adversely affect the safety of the plant in accordance with TVA procedure, NFDP-111, “Nuclear Design and Core Analysis.”

4.0 DESIGN FEATURES

4.1 Site Location

The Sequoyah Nuclear Plant is located on a site near the geographical center of Hamilton County, Tennessee, on a peninsula on the western shore of Chickamauga Lake at Tennessee River mile (TRM) 484.5. The Sequoyah site is approximately 7.5 miles northeast of the nearest city limit of Chattanooga, Tennessee, 14 miles west-northwest of Cleveland, Tennessee, and approximately 31 miles south-southwest of TVA's Watts Bar Nuclear Plant.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. Sequoyah is authorized to place a limited number of lead test assemblies into the reactor as described in the Framatome-Cogema Fuels report BAW-2328, beginning with the Unit 2 Operating Cycle 10 core.

4.2.2 Control Rod Assemblies

---NOTE---
Operation with 52 full length control rod assemblies (with no control rod assembly installed in core location H-08) is permitted during Cycle 24.

The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium, and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

b. A k_{eff} less than critical when flooded with unborated water and a k_{eff} less than or equal to 0.95 when flooded with water containing 300 ppm soluble boron. For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident; and

c. A nominal 8.972 inch center to center distance between fuel assemblies placed in the high density fuel storage racks.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water;

c. $k_{eff} \leq 0.98$ under optimum moderation conditions; and

d. The arrangement of 146 storage locations shown in Figure 4.3.1.2-1. The cells shown as empty cells in Figure 4.3.1.2-1 shall have physical barriers installed to ensure that inadvertent loading of fuel assemblies into these locations does not occur.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2091 fuel assemblies. In addition, no more than 225 fuel assemblies will be stored in a rack module in the cask loading area of the cask pit.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 342 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-328

1.0 INTRODUCTION

By letter dated April 17, 2020, as supplemented by a letter dated April 22, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML20108F672 and ML20113E939, respectively), Tennessee Valley Authority (TVA or the licensee) requested exigent changes to the Technical Specifications (TSs) for Sequoyah Nuclear Plant, Unit 2 (Sequoyah Unit 2 or SQN2). The amendment would revise Sequoyah Unit 2 TS 4.2.2, "Control Rod Assemblies," to add a note reflecting that for Unit 2 Operating Cycle 24, the core will contain 52 full-length control rod (CR) assemblies instead of 53 full-length assemblies. This change will allow Unit 2 to operate for approximately 18 months while repair plans are completed for a malfunctioning CR drive mechanism. This amendment was necessitated by emergent issues identified during CR testing conducted at the conclusion of the recent refueling outage. In addition, the licensee proposed a license condition that would require TVA to ensure the Cycle 24 core design will not adversely affect the safety of the plant in accordance with TVA procedure, NFDP-111, "Nuclear Design and Core Analysis."

TVA submitted this request under exigent circumstances to avoid further operational delays consistent with the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.91(a)(6), which state that exigent circumstances exist when a licensee and the U.S. Nuclear Regulatory Commission (NRC or the Commission) must act quickly, and time does not permit the NRC to publish a *Federal Register* notice allowing 30 days for prior public comment.

The supplemental letter dated April 22, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Chattanooga Times Free Press* on April 21, 2020.

Enclosure 2

2.0 REGULATORY EVALUATION

2.1 Description of Sequoyah Unit 2 Control Rods

As stated by the licensee in Section 3.1 of the enclosure to the license amendment request (LAR):

> SQN2 normally contains 53 full-length control rod assemblies divided into four control banks (Control Banks A, B, C, D) and four shutdown banks (Shutdown Banks A, B, C, D). Of the eight banks, Control Bank D is used for reactivity control during normal at-power operation. The remaining control banks are normally used for reactor startup and shutdown. The shutdown banks provide additional negative reactivity to meet shutdown margin (SDM) requirements. During MODES 1 and 2, the shutdown banks are fully withdrawn from the core in accordance with TS 3.1.5 and as specified in the Core Operating Limits Report (COLR).
>
> The H-08 control rod is part of Control Bank D and is located in the center of the core as shown in Figure 1 [of the enclosure to the LAR]. With the removal of the control rod in core location H-08, U2C24 will contain 52 full length control rod assemblies as shown in the table to Figure 1.
>
> Each control rod is moved by a full length CRDM [control rod drive mechanism] consisting of a stationary gripper, movable gripper, and a lift pole. Three coils are installed external to the CRDMs to electromechanically manipulate the CRDM components to produce rod motion. The CRDMs are magnetic jacking type mechanisms that move the control rods within the reactor core by sequencing power to the three coils of each mechanism to produce a stepping rod motion. Rod position is achieved through a timed sequence of stationary, movable, and lift coil current. At each point in time during rod positioning, the control rod is being held by either the stationary gripper or movable grippers.
>
> Should both sets of grippers be de-energized simultaneously, the corresponding control rod would drop into the core. The primary function of the CRDMs is to insert, withdraw, or hold control rods within the core to control average core temperature and to shut down the reactor. Mechanically, each control rod location includes a guide tube, which is an assembly that houses and guides the control rod through the upper internals.
>
> The full length Rod Control System receives rod speed and direction signals from the T_{avg} control system (contained within the Distributed Control System). The automatic rod speed demand signal varies over the corresponding range of 5 to 45 inches per minute (8 to 72 steps/minute) depending on the magnitude of the error signal. The rod direction demand signal is determined by the positive or negative value of the error signal. Manual control is provided to move a control bank in or out at a prescribed fixed speed.

Note that the terms “control rod” and “rod cluster control assemblies” (RCCAs) are used synonymously by the licensee.

2.2 Licensee Proposed Changes

Per 10 CFR 50.36(c)(4), TSs will include items in the "Design features" category where the design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in "Safety limits, limiting safety system settings, and limiting control settings," "Limiting conditions for operation," or "Surveillance requirements" categories. Sequoyah Unit 2 design feature TS 4.2.2 states:

> Control Rod Assemblies
>
> The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium, and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

The licensee proposes to revise Sequoyah Unit 2 TS 4.2.2 to add a note reflecting that for Operating Cycle 24, the core will contain 52 full-length CR assemblies instead of 53 full-length assemblies. Specifically, the licensee proposes to add the following note to TS 4.2.2:

> --NOTE---
> Operation with 52 full length control rod assemblies (with no control rod assembly installed in core location H-08) is permitted during Cycle 24.
> ---

In its April 22, 2020, supplement, the licensee proposed a license condition as follows:

> Prior to Cycle 24 startup from Unit 2 Refueling Outage 23, TVA shall ensure the Cycle 24 core design will not adversely affect the safety of the plant in accordance with TVA procedure, NFDP-111, "Nuclear Design and Core Analysis."

The licensee chose to operate without the CR assembly in Sequoyah Unit 2 core location H-08 stating that wear of the H-08 CRDM is similar to the Sequoyah Unit 1 H-08 CRDM wear, which caused the inability to maintain the CR rod in the fully withdrawn or nearly fully withdrawn position and that in-situ replacement of the affected CRDM would require special tooling that is unavailable at this time.

2.3 Regulatory Review

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. These TSs are derived from the plant safety analyses.

In Section 50.36, "Technical specifications," of 10 CFR, the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

The rule does not specify the particular requirements to be included in a plant's TSs.

The regulations in 10 CFR 50.36(c)(2)(iii)(4) state that design features are required to be in TSs to show features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety. The number of control rod assemblies is included in this section.

Licensees may propose revisions to the TSs. The NRC staff reviews proposed changes and will generally issue changes provided that the plant-specific review supports a finding of continued adequate protection of public health and safety because: (1) the change is editorial, administrative, or provides clarification (i.e., no requirements are materially altered); (2) the change is more restrictive than the licensee's current requirement; or (3) the change is less restrictive than the licensee's current requirement, but nonetheless, still affords adequate assurance of safety when judged against current regulatory standards. The detailed application of this general framework and additional specialized guidance is discussed in Section 3.0 of this safety evaluation (SE) in the context of the proposed TS changes contained in the license amendment request (LAR).

Sequoyah Unit 2 was designed to meet the intent of the Proposed General Design Criteria for Nuclear Power Plant Construction Permits published in July 1967 (Proposed GDC). The Sequoyah Unit 2 construction permit was issued in May 1970. The Sequoyah Updated Final Safety Analysis Report (UFSAR) addresses the general design criteria published as Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC), to 10 CFR Part 50 in July 1971. Each criterion is followed by a discussion of the design features and procedures that meet the intent of the criteria. Any exception to the 1971 GDC resulting from the earlier commitments is identified in the discussion of the corresponding criterion.

The GDC applicable to this LAR are listed below and discussed in detail in Section 3.4 of this SE:

GDC 2 – Design Bases for Protection Against Natural Phenomena
GDC 4 – Environmental and Missile Design Bases
GDC 10 – Reactor Design
GDC 11 – Reactor Inherent Protection
GDC 12 – Suppression of Reactor Power Oscillations
GDC 23 – Protection System Failure Modes
GDC 25 – Protection System Requirements for Reactivity Control Malfunctions
GDC 26 – Reactivity Control System Redundancy and Capability
GDC 27 – Combined Reactivity Control Systems Capability
GDC 28 – Reactivity Limits
GDC 29 –Protection Against Anticipated Operational Occurrences

While Sequoyah Unit 2 was designed to meet the intent of the Proposed GDC published in July 1967, the UFSAR addresses the GDC published as Appendix A to 10 CFR Part 50 in July 1971. Exceptions to the 1971 GDC resulting from earlier commitments are identified in the UFSAR discussions for each criterion. However, the UFSAR indicated no exceptions to the 1971 GDC used by the staff in its review.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) Section 4.6, "Functional Design of Control Rod Drive System," informed the regulatory requirements and areas of review for the proposed change.

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (SRP) Chapter 15, “Transient and Accident Analysis,” informed the regulatory requirements and areas of review for the proposed change.

3.0 TECHNICAL EVALUATION

In determining whether an amendment to a license will be issued, the Commission is guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. The NRC staff evaluated the licensee’s application to determine if the proposed changes are consistent with the regulations, guidance, and licensing and design-basis information discussed in Section 2 of this SE. The NRC staff reviewed the licensee’s statements in the LAR, the referenced NRC-approved reload methodology, and the relevant sections of the Sequoyah Unit 2 UFSAR. The NRC staff compared the impacted parameters stated in the LAR with the corresponding key safety parameters to verify that the parameters for the event were addressed and that they were bounded by the safety analysis in Chapter 15 of the UFSAR.

The licensee stated that Framatome performs the reload licensing analysis for Sequoyah Unit 2 and applies NRC-approved codes and analytical methods to design the reload core. The NRC-approved codes and analytical methods used to generate the reload SE are included in TS 5.6.3, “Core Operating Limits Report,” and are also listed in the cycle-specific COLR. The licensee stated that the reload safety analysis methods are not invalidated by the removal of CR H-08 from the Cycle 24 core design because these methods are not dependent on a particular RCCA configuration. The licensee stated that cycle-specific reload evaluations of TS limits, safety analysis limits, and operating limits without CR H-08 for Cycle 24 were performed to ensure core protective and operating limits remain satisfied and safety analysis limits remain bounded. The staff finds that the reload safety analysis methods and supporting computer codes used by the licensee remain applicable to model and evaluate the as-designed/operated configuration of the plant, as the reload methodology is not dependent upon control bank configuration.

The licensee stated that there were no changes in analytical methods or safety analysis limits used to perform the core reload SE for Cycle 24 with CR H-08 removed. The analysis supporting the evaluation of these impacted parameters was performed using the NRC-approved methodology described in TS 5.6.3.

The licensee examined impacts on margins to fuel thermal and power peaking limits related to departure from nucleate boiling and centerline fuel melt safety criteria due to the change in power distribution attributable to operation without CR H-08. The licensee also evaluated the cycle-specific power distribution maneuvering analysis to determine the acceptability of the TS and COLR operating limits related to the loss-of-coolant accident (LOCA) and loss of forced reactor coolant flow accident initial condition criteria.

3.1 Parameters Assumed in the Safety Analysis

The licensee, as part of its reload SE process, which is used for each new fuel cycle, determined the nuclear design changes and impact to core and fuel performance, as well as impact to the accident analyses described in Chapter 15 of the UFSAR for removal of CR H-08 for Cycle 24. The licensee stated that NRC-approved reload safety analysis codes and methods were used to determine if the change in core design parameters adversely impacted

the bounding key safety parameters assumed in the safety analysis. Specific parameters are discussed in the sections below. The NRC staff examined each parameter presented by the licensee to: (1) confirm the selected parameters (shutdown margin, trip reactivity, etc.) were comprehensive and consistent with the removal of CR H-08, (2) confirm that the updated values were reasonable, and (3) confirm that the safety analysis parameters as defined in the COLR and UFSAR remain bounding.

3.1.1 Shutdown Margin

Removal of a CR has a direct impact on the available shutdown margin. Shutdown margin is a requirement in Sequoyah Unit 2 TS 3.1.1 and is referenced in action statements in TSs 3.1.4, 3.1.5, 3.1.6, and 3.1.8. These TSs state that the shutdown margin shall be within the limits specified in the COLR. The Sequoyah Unit 2 COLR Section 2.1 shutdown margin limits provided in order to maintain the safety analyses, described in Chapter 15 of the UFSAR, remain bounding. The COLR specifies that for Modes 1 and 2, the shutdown margin has to be greater than or equal to 1.6 %ΔK/K. The licensee recalculated the shutdown margin after removal of CR H-08, and the value changed from 2.812 %ΔK/K to 2.253 %ΔK/K. Given the revised shutdown margin value of 2.253 %ΔK/K remains greater than the COLR limit of 1.6 %ΔK/K for Modes 1 and 2, the staff finds this acceptable.

The COLR also specifies that for Modes 3 and 4, and Mode 2 with K_{eff} < 1.0, the shutdown margin shall be greater than or equal to 1.6 %ΔK/K and must be greater than or equal to 1.0 %ΔK/K in Mode 5. The licensee stated in the LAR that the shutdown margin limits are maintained as a function of CR position and reactor coolant system (RCS) critical boron concentration for Modes 3, 4, and 5. Given the revised shutdown margin value of 2.253 %ΔK/K remains greater than the COLR limit of 1.6 %ΔK/K, the staff finds this acceptable.

While there will be a reduction in the available shutdown margin (SDM) as a result of removing CR H-08, SDM remains within the limits provided in the COLR as required by TS 3.1.1. Table 1 of the enclosure to the LAR shows that the required SDM is maintained with additional margin still available. Therefore, the staff finds that the SDM assumptions used in the safety analyses given in Chapter 15 of the UFSAR remain valid and applicable after removal of CR H-08.

3.1.2 Boron Concentration and Worth

The licensee stated in the LAR that the removal of the H-08 control rod increases the SDM boron concentration requirement to compensate for the loss in the available total RCCA negative reactivity and to compensate for the reduction in boron worth when the H-08 control rod is removed. The licensee found that the increase in the SDM boron concentration requirements in the RCS for Modes 1 through 5 ensures the removal of control rod H-08 does not impact the uncontrolled boron dilution accident.

The licensee stated that operationally, the required RCS SDM boron concentrations will be higher with the CR H-08 removed in order to meet the COLR SDM limits. Table 2 of the enclosure to the LAR provides the minimum required shutdown boron concentration with all rods in minus the most reactive stuck rod for 1.6% SDM and 1.0% SDM for beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) conditions.

Therefore, the staff finds that with the removal of CR H-08, the boron concentration change based on meeting SDM requirements is still within the assumed values for the UFSAR events

and does not impact the results presented in UFSAR Section 15.2.4, “Uncontrolled Boron Dilution.”

3.1.3 Trip Reactivity

The removal of CR H-08 reduces the trip reactivity as a function of rod insertion position, which reduces the trip reactivity as a function of time after the CRs begin to fall. The normalized trip reactivity as a function of CR insertion position and normalized trip reactivity as a function of time after the CRs begin to fall is presented in the UFSAR in Figures 15.1.5-2 and 15.1.5-3. The licensee stated that the curve of trip reactivity as a function of time used in the safety analyses is verified to be bounding by a cycle-specific calculation of the minimum trip worth at hot full power (HFP) and hot zero power (HZP). An evaluation of the effects of the removal of CR H-08 was performed by the licensee. The results show that while the minimum trip worth was reduced when CR H-08 is removed, the minimum trip worth is greater than the limit of 4,000 percent mille, and therefore, the curve of trip reactivity as a function of time after the CRs begin to fall used in the safety analyses remains bounding. Therefore, the staff finds that the removal of CR H-08 does not impact the trip reactivity assumed in UFSAR Chapter 15 events.

3.1.4 Moderator Temperature Coefficient

The moderator temperature coefficient (MTC) is one of the controlling parameters for core reactivity in both overheating and overcooling accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the safety analyses consider worst case conditions to ensure that the accident results are bounding. The consequences of accidents that cause core overheating must be evaluated with the most positive MTC. Such accidents include the uncontrolled bank withdrawal transient from any power level, loss of electrical load, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated with the most negative MTC. Such accidents include sudden feedwater flow increase and steam line breaks.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The LAR states:

> The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC [beginning of cycle] or EOC [end of cycle] condition. The most conservative combination appropriate to the accident is then used for the analysis.

The values used in the safety analysis are summarized in UFSAR Table 15.1.2-2.

The removal of CR H-08 slightly impacts the MTC calculated at the conservative bounding conditions determined for the UFSAR accident analyses. The LAR presents MTC results for cases with and without CR H-08. The NRC staff compared the values against the limits provided in the LAR, the values in the COLR, and the values used in the UFSAR Chapter 15 analysis and found that the values used in the UFSAR remain conservative.

3.1.5 Miscellaneous Safety Analysis Neutronic Parameters

The licensee stated that miscellaneous safety analysis neutronic parameters such as delayed neutron data (beta and prompt neutron lifetime), Doppler temperature coefficients, and fuel temperatures are not significantly impacted by the change in core configuration. These

parameters are driven more directly by the core design and not the CRs. The staff finds this acceptable, given that the licensee's cycle-specific parameter evaluations of these safety analysis values show negligible changes and confirm that the values assumed in the safety analysis remain bounding.

3.2 Impact on UFSAR Chapter 15 Accident Analysis

As stated by the licensee, removal of CR H-08 has an impact on most comparisons to UFSAR Chapter 15 accident analysis parameters routinely evaluated as part of the reload SE. In addition to the items discussed above in Section 3.1 (SDM, MTC, trip reactivity, boron concentration, etc.), the impact of removal of CR H-08 on control rod worth, margin to peaking limits (departure from nucleate boiling (DNB), and centerline fuel melt (CFM)), and other accident analysis parameters was also evaluated by the licensee.

The licensee performed cycle-specific evaluations to determine if the change in core configuration adversely impacts bounding key safety parameters assumed in the UFSAR Chapter 15 safety analysis and any impacts on DNB and fuel thermal margins due to the change in power distribution. Table 14 of the enclosure to the LAR provides the impact on the UFSAR Chapter 15 accident analyses due to the removal of CR H-08. The licensee's table provides the UFSAR section, description of the event, and the licensee's comment on the accident impact. For the majority of cases, the licensee stated that the removal of CR H-08 has no effect on the analysis of record and that cycle-specific evaluations verify the analysis of record remains bounding. There were a few events where removal of CR H-08 has no impact. These include events such as steam generator tube rupture (CRs not explicitly modeled), fuel handling accident (no relevant analysis parameters affected), and waste gas decay tank rupture (no relevant analysis parameter affected). Overall, the licensee found that the removal of CR H-08 for Cycle 24 does not impact the results presented in UFSAR Chapter 15. A summary of the licensee's findings as stated in the application is below.

- UFSAR Chapter 15 accidents with rod worth limits show the available Control Bank D worth for drop/insertion/withdrawal will be less due to removal of H-08 control rod from the U2C24 core.

- REA [rod ejection accident] ejected rod worths increased for the HFP cases and were reduced for HZP cases. Peaking results were reduced for all cases with the exception of the HFP BOC case, which increased but maintained margin to the safety limit. These changes are due to the power shifting more towards the center of the core during the REA due to the removal of the control rod in core location H-08. This power distribution change reduced the peaking for the REA for most cases and changed the ejected rod worths only slightly. REA bounding initial conditions assumption for the safety analysis remain unchanged for this cycle, and without a control rod in core location H-08, an REA will not occur in the H-08 location.

- [Dropped rod accident] DRA DNB [departure from nucleate boiling] margins decreased due to the removal of the H-08 control rod, but still had significant margin to the limit. The DRA Control Bank D rod group previously containing H-08 will go from 5 to 4 control rods with H-08 excluded.

- HZP SLB [steam line break] peaking results were reduced because power was anchored toward center of core with no control rod in H-08 and maximum stuck rod out. The reduction in peaking of the No H-08 case relative to the rodded H-08 case meant the rodded H-08 DNB and CFM [centerline fuel melt] calculations bounded the No H-08 case, and therefore, the No H-08 DNB and CFM margin calculations were not necessary. The HZP SLB reactivity decreased versus safety analysis limits due to the absence of the H-08 control rod and had significant margin to the limit.

- The SLB c/w RWAP [bank withdrawal at power] parameter evaluation results with no control rod in H-08 were less limiting than the rodded H-08 evaluation. Statepoint reactivity differences resulting from the bank withdrawal decreased relative to the limit due to the absence of H-08 from the withdrawn Control Bank D. Safety analysis required MTC ranges continued to be satisfied with the removal of the H-08 control rod.

- The increase in SDM boron concentration requirements ensures the UBDA [uncontrolled boron dilution accident] for modes 1 through 5 remains bounding for the removal of the H-08 control rod.

- Single Rod Withdrawal accident no change in the number of failed fuel pins in the pin census.

- Uncontrolled Control Bank Withdrawal (UCBW) at power saw reduced maximum reactivity insertion rates due to the absence of the H-08 control rod from Control Bank D and increased margins to the safety analysis limits.

- UCBW from subcritical saw the maximum reactivity insertion rate decrease due to the removal of the H-08 control rod. However, maximum radial pin power increased because the absence of the H-08 control rod allowed power to move strongly to the center of the core. The higher calculated maximum radial pin power with the H-08 control rod removed satisfied the limit.

- SDM and maximum insertable worth were reduced due to removal of control rod in H-08 with subsequent reduction in available rod worth; however, adequate margin to the SDM limit remains.

The staff finds the licensee’s use of the NRC-approved reload SE methods described in TS 5.6.3 acceptable and appropriate to determine if the removal of CR H-08 adversely impacts the bounding key safety parameters assumed in the UFSAR Chapter 15 safety analysis and the impacts on DNB and CFM, due to the change in power distribution attributable to the new core design. The licensee reasonably appears to have examined all pertinent parameters for each transient. Given that the licensee’s cycle-specific parameter evaluations for UFSAR Chapter 15 safety analysis parameters confirm that the values assumed in the safety analysis remain bounding for all UFSAR Chapter 15 safety analysis accidents, the staff finds the removal of CR H-08 acceptable for Cycle 24.

By letter dated April 22, 2020, the licensee stated that during the offload and inspection of fuel assemblies for Sequoyah Unit 2 Cycle 24, fuel assembly ML04 was found to be damaged. It was determined by the licensee that fuel assembly ML04 cannot be loaded as planned. This

has necessitated the core design being modified from that documented in the LAR. The licensee stated that the redesigned core is neutronically similar and physically identical to the core described in the LAR with the following changes: (1) the assembly slated for use in core location M-09 (fuel assembly ML04) is discharged to the spent fuel pool, (2) the assembly slated for use in core location R-08 is moved to core location M-09, and (3) an assembly from the spent fuel pool is in core location R-08.

The Sequoyah fuel vendor has established redesign criteria to assess differences between core designs to determine if substantial changes to key parameters will challenge safety analysis limits. The licensee stated that changes to the core design described in the LAR and the redesigned core acceptably met the redesign criteria, requiring only a few core and safety analysis parameters to be recalculated. With the reduced scope core redesign process being used, the licensee determined that most of the safety analysis parameters described in the LAR do not need to be recalculated to ensure the redesigned core meets safety analysis limits and that the proposed license condition that will utilize existing TVA procedures will ensure the redesigned Sequoyah Unit 2 Cycle 24 core without CR H-08 meets all safety analysis acceptance criteria.

The NRC staff acknowledges that some of the parameters presented in the LAR and described above in Section 3.1 may change due to the redesigned core. Given that only two fuel assemblies (out of 193) have been changed, the staff would expect only very minor changes to the parameters discussed above in Section 3.1, and there would still be margin to the acceptance criteria. In addition, in the April 22, 2020, supplement, the licensee proposed a license condition to confirm that the redesigned core will operate within the bounds of the safety analyses. The proposed license condition states that the licensee will follow TVA procedure, NFDP-111, "Nuclear Design and Core Analysis," to ensure the Cycle 24 core design will not adversely affect the safety of the plant. The NRC staff finds that use of the reduced scope core redesign process is an appropriate method to ensure the redesigned core meets safety analysis limits and that following existing procedures and analysis methods will ensure the redesigned Sequoyah Unit 2 Cycle 24 core will operate within the bounds of the safety analyses without CR H-08.

3.3 Impact of the Flow Restrictor

3.3.1 Thermal-Hydraulic Impacts

The licensee stated that when CR H-08 and its associated driveshaft are removed from service, a flow restrictor will be installed in the H-08 CR guide tube in the reactor vessel upper internals. Installation of the flow restrictor will ensure the flow area and hydraulic resistance normally provided by the driveshaft in the guide tube will be maintained. The licensee performed a bypass flow analysis to determine the impact of removing the CR in core location H-08. The licensee stated that this analysis shows that the core bypass flow increases slightly but remains below the analyzed bounding value, and therefore, all DNB analyses remain bounding.

In addition, the increase in the core bypass flow has the potential to affect the system transient analyses, and a disposition of events was performed for the UFSAR Chapter 15 events. The bypass flow is a less significant parameter in the system analyses than it is in the DNB analyses. Framatome determined that the existing large and small break LOCA analyses remain bounding. Furthermore, the non-LOCA UFSAR Chapter 15 accident analyses continue to be applicable, considering the incremental increase in bypass flow due to the removal of CR H-08.

The NRC staff concludes that the licensee's proposed installation of a flow restrictor is acceptable because the flow restrictor maintains the thermal-hydraulic configuration of the reactor vessel upper internals.

3.3.2 Structural Evaluation

3.3.2.1 Dynamic Analysis

The licensee stated that the changes in RCS water volume and metal mass is negligible due to removal of the CR and the installation of the flow restrictor. The licensee concluded that the effect on the dynamic analysis is negligible. The NRC staff has determined that the change in system masses is negligible. Therefore, the staff finds that the impact on the dynamic analyses that predicts the stresses in the CRDM, reactor vessel, vessel supports, and reactor internals when subjected to seismic or LOCA excitations is negligible.

The LAR states that there is no impact on the functionality or structural integrity of the reactor vessel upper internals with the removal of the control rod drive shaft and RCCA at core location H-08 when a flow restrictor is installed in its place. Therefore, there is no impact on the current reactor vessel internals analyses. The NRC staff agrees that there should not be significant impact on the current reactor vessel internals analysis because the reactor internals hydraulic loads will be similar, as the flow restrictor provides similar flow and pressure loss at core location H-08.

3.3.2.2 Flow Restrictor Design

The licensee stated that the installed flow restrictor is a standard component used to hydraulically simulate the CRDM drive shaft clearance with the guide tube housing opening. This will establish hydraulically equivalent flow conditions in the upper internals when the drive shaft is removed. The licensee performed a generic structural analysis of the restrictor plate/orifice assembly using a bounding pressure differential load for the faulted service condition (loss-of-coolant accident or LOCA). This analysis conservatively assumed no orifice holes in the assembly to maximize the differential pressure load. The analysis demonstrated that all membrane and bending, bearing, and shear stress intensities satisfy the requirements of the 1989 Edition of the ASME Boiler and Pressure Vessel Code (BPV Code), Section III. The licensee also demonstrated that bolting preload was adequate to resist assembly separation for maximum LOCA pressure loads. The NRC staff agrees that the generic analysis bounds the Sequoyah Unit 2 plant-specific service conditions.

Further, the NRC staff notes that the materials used for the flow restrictor assembly conform to the ASME BPV Code, Section II, Part A. The licensee indicated that the material is Type 304 stainless steel for restrictor assembly, as well as the guide tube, and is compatible with fluid conditions in the reactor vessel upper internals. The NRC staff agrees that there will be no differential thermal expansion effects because the restrictor assembly and the guide tube are both of the same material.

The licensee also discussed the design to prevent the possibility of loose parts in the RCS. Installation of the restrictor is controlled to ensure that the required hex bolt preload is obtained, securely locking the flow restrictor in place at the top of the guide tube. A locking cup, which is tack welded to the flow restrictor, is crimped onto the hex bolt to prevent hex bolt rotation. The NRC staff agrees that the capture features of the flow restrictor (i.e., locking fingers, hex bolt

cup, and hex bolt preload) provide reasonable assurance that the flow restrictor is securely installed and will not result in the generation of loose parts.

The LAR stated that the reactor internals at Sequoyah Unit 2 are designed and analyzed to the requirements of UFSAR Section 3.9.3, "NSSS Components Not Covered by the ASME Code." The basis for the design stress and deflection criteria is summarized in Section 4.2.2.5 of the UFSAR. While the restrictor assembly does not perform a core support or safety function, it is classified as ANSI Safety Class 3. All of the calculated stresses are within the ASME BPV Code allowable stress limits. The NRC staff has determined that the flow restrictor assembly materials, fabrication, and design analysis meet the intent of ASME BPV Code, Subsection NG, consistent with the Sequoyah Unit 2 design basis per the UFSAR.

Based on review of the LAR, the NRC staff has determined that there is reasonable assurance that the flow restrictor will maintain its structural integrity without generating loose parts as ASME BPV Code materials and the design allowable stress limits in accordance with the Code are utilized.

3.4 Compliance with Applicable GDC

The staff's review of the applicable GDC listed in Section 2.3 above is summarized below.

GDC 2 - Design Bases for Protection Against Natural Phenomena

> Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety function. The design bases for these structures, systems, and components shall reflect:
>
> 1. Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated,
>
> 2. Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and
>
> 3. The importance of the safety functions to be performed.

The NRC staff finds that the current CRDM dynamic stress evaluations due to seismic and LOCA excitations remain valid because the impact of the mass change or reduction is not significant.

GDC 4 - Environmental and Dynamic Effects Design Bases

> Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids,

> that may result from equipment failures and from events and conditions outside the nuclear power unit.

The licensee evaluated the removal of the CR assembly and the addition of the flow restrictor. The existing analyses for most RCS subcomponents remain bounding, and the stresses associated with the flow restrictor are within the ASME Code limits. Further, there will be no differential thermal expansion effects, and the capture features of the flow restrictor (i.e., locking fingers, hex bolt cup, hex bolt preload) provide reasonable assurance that the flow restrictor is securely installed and will not result in the generation of loose parts. The NRC staff has determined that the capture features provide reasonable assurance that the flow restrictor will not result in the generation of loose parts. The NRC staff finds that this meets the requirements of GDC 4, because acceptable component design limits will not be exceeded during normal operation, including the effects of anticipated operational occurrences.

GDC 10 – Reactor Design

> The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The licensee performed a redesign reload analysis in accordance with the methods described in TS 5.6.3 and confirmed that the fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences with CR H-08 removed. The staff finds that this meets the requirements of GDC 10, since acceptable fuel design limits will not be exceeded during normal operation, including the effects of anticipated operational occurrences.

GDC 11 – Reactor Inherent Protection

> The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

The fuel temperature coefficient is negative, and the moderator temperature coefficient of reactivity is non-positive for power operating conditions, thereby providing negative reactivity feedback characteristics. The NRC staff finds that this criterion remains satisfied because removal of CR H-08 does not impact the ability to detect or control core power distribution, and the at-power nuclear reactivity feedback coefficients remain unchanged.

GDC 12 – Suppression of Reactor Power Oscillations

> The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

The licensee states that power oscillations of the fundamental mode are inherently eliminated by the negative Doppler and non-positive moderator temperature coefficient of reactivity. Oscillations due to xenon spatial effects in the radial, diametral, and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative Doppler and nonpositive

moderator temperature coefficients of reactivity. Oscillations due to xenon spatial effects in the axial first overtone mode may occur. Assurance that fuel design limits are not exceeded by xenon axial oscillations is provided as a result of reactor trip functions using the measured axial power imbalance as an input. Oscillations due to xenon spatial effects in axial modes higher than the first overtone are heavily damped due to the inherent design and due to the negative Doppler coefficient of reactivity.

The NRC staff finds that this criterion remains satisfied, as the safety analysis with CR H-08 removed demonstrates that it will not result in power oscillations, which would result in conditions exceeding specified acceptable fuel design limits.

<u>GDC 23 – Protection System Failure Modes</u>

> The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

The overall design of the protection system was not changed. The removal of CR H-08 from the reactor vessel does not impact the fail-safe function of the remaining 52 CRs, which will still reliably maintain an adequate reactor shutdown capability. The physical removal of the CR drive shaft does not have any mechanical impact on the function of the remaining 52 CRs. The electrical removal from service of CR H-08 involves pulling fuses to remove control power to the respective stationary, lift, and movable coils. The remaining CRs are not impacted by this electrical change and will continue to meet their design function. The licensee's modification design change process ensures that the associated plant modifications involve only CR H-08 and do not affect other CRs.

The NRC staff finds that this criterion remains satisfied by maintaining the CR insertion capability with the remaining 52 CRs.

<u>GDC 25 – Protection System Requirements for Reactivity Control Malfunctions</u>

> The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

The protection system is designed to limit reactivity transients so that fuel design limits are not exceeded. UFSAR Chapter 15 events were confirmed to be bounding for analyzed malfunctions of the reactivity control systems.

The NRC staff finds that this criterion remains satisfied as the reactor trip function remains fully capable of performing its function with 52 CRs, and fuel design limits were not exceeded for analyzed malfunctions of the reactivity control systems with the removal of CR H-08.

<u>GDC 26 – Reactivity Control System Redundancy and Capability</u>

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling

> reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Two reactivity control systems are provided, including the rod cluster control assemblies and chemical shim (boration). The RCCAs are inserted into the core by the force of gravity. The boron chemical shim is unaffected and will maintain the reactor in the cold shutdown state, independent of the position of the CRs, and can compensate for all xenon burnout transients.

The NRC staff finds that this criterion remains satisfied, as the licensee's analysis has demonstrated that removal of CR H-08 does not impact the ability of the reactivity control system to perform its function. Under normal operating conditions, including anticipated operational occurrences, acceptable fuel design limits were demonstrated to not be exceeded.

GDC 27 – Combined Reactivity Control Systems Capability

> The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

The reactivity control is achieved by a combination of RCCA and automatic boron addition via the emergency core cooling system with the most reactive CR assumed to be fully withdrawn. Manually controlled boric acid addition is used to supplement the RCCA in maintaining the SDM for the long-term conditions of xenon decay and plant cooldown.

The NRC staff finds that this criterion remains satisfied with the removal of CR H-08, as the licensee's analysis has demonstrated that the ability of the reactivity control systems to reliably control reactivity changes and that adequate SDM is maintained when considering the highest stuck rod worth. The licensee's evaluations of the removal of CR H-08 during Cycle 24 demonstrate that SDM and safety analysis limits are met throughout the fuel cycle.

GDC 28 – Reactivity Limits

> The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

The appropriate reactivity insertion rate for the withdrawal of RCCA and the dilution of the boric acid are controlled by the TSs. The specification includes or references appropriate graphs that show the permissible mutual withdrawal limits and overlap of functions of the several RCCA banks as a function of power.

The NRC staff finds that this criterion remains satisfied, as the licensee's analysis with removal of CR H-08 demonstrates trip reactivity insertion rate, SDM, and the safety analysis limits remain met for the UFSAR Chapter 15 accidents for the entire fuel cycle (Cycle 24).

GDC 29 – Protection Against Anticipated Operational Occurrences

> The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

The licensee stated that the protection and reactivity control systems are designed to ensure an extremely high probability of fulfilling their intended functions. The design principles of diversity and redundancy, coupled with a rigorous quality assurance program and analyses, support this probability, as does operating experience in plants using the same basic design.

The NRC staff finds that this criterion remains satisfied, as the removal of CR H-08 does not impact the ability of the reactivity control systems to perform their safety functions. The mechanical removal of the CR drive shaft and RCCA do not have any mechanical impact on the function of the remaining 52 CRs. The remaining 52 CRs are also not impacted by the related electrical changes when CR H-08 is removed. Therefore, the staff finds that a high probability CR insertion continues to exist under anticipated operational occurrences, even with the removal of the H-08 CR during Cycle 24.

4.0 RENEWED FACILITY OPERATING LICENSE CHANGE

Renewed Facility Operating License No. DPR-79 is amended by addition of License Condition 2.C(28) as follows:

> (28) Prior to Cycle 24 startup from Unit 2 Refueling Outage 23, TVA shall ensure the Cycle 24 core design will not adversely affect the safety of the plant in accordance with TVA procedure, NFDP-111, "Nuclear Design and Core Analysis."

The NRC staff's review of this proposed license condition can be found in Section 3.2 of this SE.

5.0 TECHNICAL CONCLUSION

The licensee proposed to modify TS 4.2.2 to add a note stating that operation with 52 full-length CR assemblies (with no CR assembly installed in core location H-08) is permitted during Cycle 24. The NRC staff has reviewed the results of the reload SE process for use in justifying the specific circumstances addressed in this LAR (e.g., Cycle 24 of Sequoyah Unit 2, the location of the rod in the core, the rod worth, the moderator density coefficient and the impact on the accident analyses, etc.) The licensee has not requested permanent removal of CR H-08. As such, a full scope review of the licensee's UFSAR Chapter 15 events is not warranted or needed to make a safety determination for one operating cycle. Additional plant design changes under different circumstances for different cycles may require additional staff review.

The NRC staff concludes that the licensee's proposed use of 52 CR assemblies in Sequoyah Unit 2 for Cycle 24, including the use of a flow restrictor, is acceptable because the design change is consistent with the current design basis and does not challenge the safety analyses detailed in Chapter 15 of the UFSAR. Since a licensee's TSs are derived from its UFSAR analyses, and the proposed change does not adversely affect the licensee's UFSAR analyses or change any applicable TS limiting condition for operation and surveillance requirements, the staff finds the change acceptable per the requirements in 10 CFR 50.36. The staff concludes that the licensee used methods consistent with regulatory requirements and guidance identified in Section 2.0 above. The NRC staff also finds the proposed use of 52 CR assemblies continues to meet the requirements of GDC 2, 4, 10, 11, 12, 23, 25, 26, 27, 28, and 29.

6.0 EXIGENT CIRCUMSTANCES

The NRC's regulations contain provisions for issuance of amendments when the usual 30-day public comment period cannot be met. These provisions are applicable when both exigent circumstances exist and the amendment involves no significant hazards consideration. Consistent with the requirements in 10 CFR 50.91(a)(6), exigent circumstances exist when a licensee and the NRC must act quickly, and time does not permit the NRC to publish a *Federal Register* notice allowing 30 days for prior public comment. As discussed in the licensee's application, the licensee requested that the proposed amendment be processed by the NRC on an exigent basis.

In its April 17, 2020, LAR, TVA provided the following timeline and justification related to CR H-08:

> SQN1 was initially unable to startup from the U1R23 outage in November 2019 due to unreliable RCCA performance in core location H-08. The control rod in core location H-08 was unable to be reliably held in the withdrawn position due to worn stationary gripper latch mechanisms in the control rod drive system, and unexpectedly dropped three times during Mode 3 testing activities at the end of U1R23. The SQN1 H-08 control rod was unable to maintain a withdrawn position reliably, and was unlikely to remain in position for the entire cycle.
>
> The replacement of a CRDM of similar design and installation configuration had not been performed in the United States. In-situ replacement of the H-08 control rod CRDM required specially modified tooling, similar in nature to the original manufacturing tooling, which did not currently exist. Additionally, the planning associated with a CRDM replacement activity would have required fabrication of mockups to test the effectiveness of the tooling, methods, and procedures. The planning and preparation process was expected to require a lead time on the order of months. Therefore, the above repair/replacement discussion provided the basis for exigency.
>
> Consideration was given to operating SQN1 during U1C24 with the H-08 control rod fully inserted in the core. This option was not considered to be viable for the following reasons:
>
> - The core would be susceptible to radial xenon oscillations that would challenge operator responses,

- Uneven depletion of fuel assemblies would have had a significant impact on the core design for future fuel cycles with regard to safety/operating margins and fuel economy, and
- The impact on core power distribution would likely require operation at a reduced power level.

TVA determined that the safest option was to operate SQN1 during U1C24 with the H-08 control rod removed. Based on the above, this option was unavoidable and was exigent in nature.

In response to the SQN1 operating experience, an extent of condition exam was planned for the U2R23 refueling outage. A sampling of 14 CRDM locations was selected for the scope of this inspection. This sample population included H-08, the eight surrounding locations, a subset of control bank "D", and locations in shutdown banks in order to best understand which locations are most susceptible to this wear mechanism. A set of examination criteria was developed based on all available data to assess the potential risk that SQN2 CRDMs may be in the same material condition as the SQN1 H-08 CRDM. This included a review of CRDM coil current trace data, rod drop times, guide card wear, and thermal sleeve wear data. With this supporting information, an inspection procedure and decision flow chart was prepared by TVA and the original equipment manufacturer. This material directed that any CRDM with a latch arm tip thickness worn to zero thickness (as was observed in the SQN1 H-08 CRDM inspections) and also had unexpected CRDM coil trace results would be graded as a high risk of a control rod drop. The SQN2 H-08 CRDM met both of these criteria, a condition that could not have been known until inspection results were obtained during the outage.

Inspections were completed for all 14 CRDM locations selected for the U2R23 refueling outage examination scope. The 13 remaining locations were inspected, reviewed, and determined to have adequate latch arm tip thickness with minimal wear reported. Based on the inspection and review, the risk condition of all remaining inspected CRDM stationary gripper latch mechanisms was determined to be low based on the examination criteria discussed previously.

Because the condition of the SQN2 CRDM stationary gripper latch mechanisms was not completely known pre-outage, reasonable preparations were made with the original equipment manufacturer in the event that CRDM repair or replacement would have been shown to be the best option. Many of the same challenges regarding domestic precedence, tooling, qualifications, and implementation that existed at the time of the SQN1 H-08 operating experience still exist today. A repair method has also been considered, which would replace only the internal latch assembly instead of the whole CRDM. While this slightly reduces the scope of work required, implementation schedules and challenges remain similar to the CRDM replacement option.

In summary, the extent of condition examination of the SQN2 H-08 control rod revealed similar wear indications on the CRDM stationary gripper latch mechanism. It is therefore believed that SQN2 H-08 control rod will not reliably maintain in the withdrawn position, resulting in SQN2 being unable to startup or maintain power operation. As with SQN1, TVA has determined that the safest

option is to operate SQN2 during U2C24 with the H-08 control rod removed. This option is similarly unavoidable and is exigent in nature. TVA plans to provide the technical evaluation supporting permanent removal of the SQN2 H-08 control rod in a future license amendment request.

Summary

The NRC staff confirmed the above circumstances onsite and finds that the licensee made a timely application for the proposed amendment following identification of the issue. In addition, the NRC staff finds that the licensee could not avoid the exigency because the condition the licensee found for SQN2 related to CR H-08 had not been identified previously by the licensee, the original equipment manufacturer, or in industry, and the licensee acted quickly upon discovery of the condition. Therefore, the licensee has not failed to use its best efforts to make a timely application for an amendment in order to create an exigency and take advantage of the exigency procedure in 10 CFR 50.91(a)(6). Based on these findings and the determination that the amendment involves no significant hazards consideration as discussed in Section 7.0 below, the NRC staff has determined that a valid need exists for issuance of the license amendment using the exigent provisions of 10 CFR 50.91(a)(6).

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

Under the provisions in 10 CFR 50.91(a)(6), the NRC notifies the public in one of two ways: (1) by issuing a *Federal Register* notice providing notice of an opportunity for hearing and allowing at least 2 weeks from the date of the notice for prior public comment or (2) by using local media to provide reasonable notice to the public in the area surrounding the licensee's facility. In this case, a notice was published on April 21, 2020, in the *Chattanooga Times Free Press* requesting comment by 4:00 p.m. on April 23, 2020.

As required by 10 CFR 50.91(a)(1), when a licensee requests an amendment, it must provide to the Commission its analysis about the issue of no significant hazards consideration using the standards in 10 CFR 50.92. Under 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

The licensee's determination of no significant hazards consideration is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   Operation of SQN, Unit 2, Cycle 24 with the H-08 control rod removed will not involve a significant increase in the probability or consequences of an accident previously evaluated. Shutdown Margin (SDM) is reduced by the absence of the H-08 control rod, but remains bounded by the limits specified by the Core Operating Limits Report (COLR). Because the impacts on the cycle-specific nuclear design parameters are bounded by the conservative input values used in the Updated Final Safety Analysis

Report (UFSAR) accident analyses, the current accident analyses remain bounding. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   Operation of SQN, Unit 2, Cycle 24 with the H-08 control rod removed will not create the possibility of a new or different kind of accident from any accident previously evaluated and the safety evaluations performed for U2C24 with the H-08 control rod removed validated that the impacts to the nuclear design parameters were within the bounds of those already assumed in the UFSAR Chapter 15 accident analyses. The current accident analyses remain bounding. Additionally, by installing a flow restrictor in the H-08 upper internals control rod guide tube, the hydraulic characteristics of the reactor vessel upper internals hydraulic characteristics are unchanged and all plant equipment will continue to meet applicable design and safety requirements. Therefore, the proposed change does not create the possibility of a new or different kind of accident than those previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   Operation of SQN, Unit 2, Cycle 24 with the H-08 control rod removed will not involve a significant reduction in a margin of safety. The margin of safety is established by setting safety limits and operating within those limits. The proposed change does not alter any UFSAR design basis or safety limit and does not change any setpoint at which automatic actuations are initiated. The proposed change has been evaluated for effects on available shutdown margin, boron worth, trip reactivity as a function of time, and moderator temperature coefficient. The results of these evaluations show that the proposed change does not exceed or alter a design basis or safety limit. Therefore, the proposed change does not significantly reduce a margin of safety.

The licensee stated in its supplement of April 22, 2020, that the proposed license condition does not change its determination of no significant hazards consideration. Based on the above evaluation, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has determined that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

8.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment on April 22, 2020. The State official did not provide comments.

9.0 PUBLIC COMMENTS

On April 21, 2020, in the *Chattanooga Times Free Press*, the NRC staff published a public notice associated with the proposed amendment request. In accordance with the requirements in 10 CFR 50.91 for an exigent amendment, the notice provided until 4:00 p.m. April 23, 2020, for public comment on the proposed no significant hazards consideration determination. No comments were received.

10.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, published in the *Chattanooga Times Free Press* on April 21, 2020, and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

11.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: R. Beaton
J. Wilson
R. Grover
J. Kaizer
C. Basavaraju
Y. Wong

Date: April 23, 2020

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNIT 2 - ISSUANCE OF EXIGENT AMENDMENT NO. 342 TO OPERATE ONE CYCLE WITH ONE CONTROL REMOVED (EPID L-2020-LLA-0078) DATED APRIL 23, 2020

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